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An ITER-like wall for JET

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Abstract

This article presents an overview of the new ITER-like wall project in JET. It aims at an optimal use of JET's unique features: physical size, plasma parameters most closely to ITER and the capability to handle beryllium and tritium, allowing the study of critical questions related to operating within the limits of the ITER wall materials. A full replacement of the first wall materials is planned (beryllium in the main wall and tungsten in the divertor). This should deliver answers to urgent plasma surface interaction questions such as tritium retention and provide operational experience in steady and transient conditions with ITER wall materials under relevant geometry and relevant plasma parameters. In addition, the JET auxiliary heating power will be upgraded to ~45 MW, allowing access to ITER-relevant disruption and edge localised modes energy loss densities. This will open access to conditions of melt layer formation both on the beryllium first wall and the tungsten divertor.

PACS: 52.55.Fa

Keywords: JET; First wall; Divertor material; Beryllium; Tungsten

1. Introduction

The construction of the International Tokamak Experimental Reactor (ITER) has been decided early 2006. The design of ITER is the collaborative result of decades of fusion research from world-wide coordinated fusion activities resulting in a large and robust database giving a good confidence to achieve the scientific and technological objectives within appropriate margins. However, a set of critical issues remains for which the present database is less robust and which determine the ongoing fusion research in present devices. These issues are largely related to plasma–wall interaction processes such as the control of the in-vessel tritium (T) inventory, the control of the wall and divertor power loads

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below technically acceptable limits and to achieve a sufficient lifetime of the plasma facing components (PFCs). These questions are of less importance in present devices but will dominate operational constraints in ITER and beyond, where the optimisation of the core plasma performance and wall erosion, tritium retention and lifetime issues cannot be decoupled. As shown in Fig. 1(a), ITER is currently designed to have an all-beryllium-clad first wall, tungsten (W) brushes over most of the divertor region and carbon fibre composites (CFC) only at the target plates where the highest power fluxes are expected (divertor strike points) [1]. The ITER reference materials have been tested in isolation in tokamaks, plasma simulators, ion beams and high heat flux test beds. However, an integrated test demonstrating both acceptable T retention and the ability to operate ITER-relevant plasmas with high power input within the limits set by these materials has not yet been demonstrated. Within the 'JET programme in support of ITER', the objective of the ITER-like wall project is to install in JET, a beryllium (Be) wall and an all-W divertor (the favoured back-up materials solution for ITER, see Fig. 1(b)). This choice is more demanding than the ITER-reference combination but offers a cleaner comparison between an all-metal and an all-CFC JET wall; it will also provide relevant information for the preparation of an all-W divertor option on ITER.

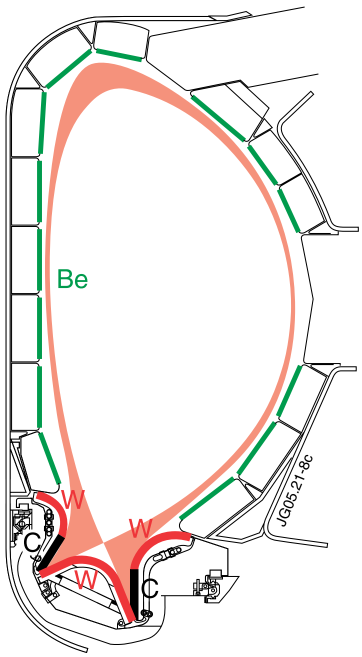

Fig. 1(a). ITER primary materials choice – backup solution is an all tungsten divertor.

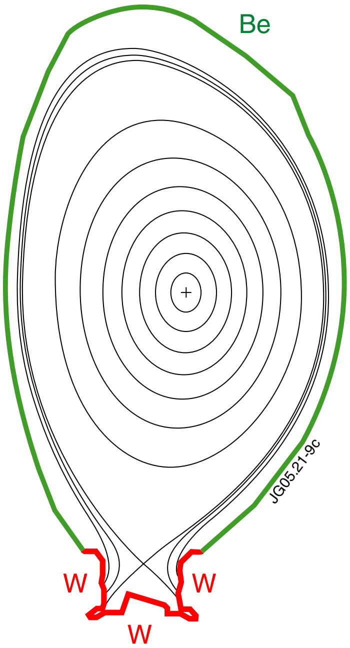

Fig. 1(b). JET first priority is to test an all W divertor – matches ITER fall back or second phase option.

The main constraints on the ITER-like wall project are the needs to preserve the power, energy handling and force limits (due to disruptions) set for the CFC wall whilst providing a scientifically relevant materials configuration for ITER. These constraints have led to the design of solid Be tile which are inertially cooled, segmented (to minimise eddy forces) and castellated (to avoid thermal stress cracking) with hidden bolts and tile shaping to maximise the power handling. These aims have had to be compromised with the technical capabilities of the existing supporting wall and limiter structures since a redesigning of these structures is beyond the scope and resources of the project. Main wall elements in contact with the plasma will be made from solid Be whilst larger areas of recessed inner wall cladding will be made from Be-coated inconel tiles. W-coated CFC tiles will be used on recessed areas of the neutral beam injection (NBI) shine through where the power handling capability of Be was insufficient. For the divertor, an extensive R&D programme has been completed in which 14 different types of W coatings on CFC have been tested on JET-relevant 2D CFC sample tiles. In addition, the concept for a single row of bulk tungsten tiles for the outer divertor has been developed. To meet the schedule, a major upgrade of remote handling systems and new tooling are required to install the novel tile designs, but this is not part of the present paper.

2. ITER-like wall in JET: scientific overview

A main concern with the ITER materials choice is the long term T retention behaviour [2–4]. In carbon-clad devices this is mainly caused by erosion of carbon and its final migration to plasma shadowed areas and co-deposition with hydrogen isotopes. Present estimates show that ITER would reach its permitted 350 g T inventory in relatively few pulses (probably ~100 full performance pulses). On the other hand, current T removal techniques are not elaborated enough to provide a solid basis on which ITER could operate. This is the main reason why an all-W divertor has been considered as the alternative option for the deuterium–tritium phase of ITER. However, the use of W and Be has other serious implications for ITER, mainly with respect to Be and W melting, mixed Be–W material issues and W core plasma contamination. The JET experiment can significantly increase the confidence in the installation of an all-tungsten divertor in ITER. The objectives of the JET experiment are as follows:

- Demonstrate that a Be wall plus an all-W divertor have sufficiently low fuel retention to meet ITER requirements.
- Demonstrate ITER-relevant T retention mitigation and detritiation techniques in a Be/W machine – including in particular the effect of trace oxygen and carbon impurities on fuel retention.
- Show how Be migration onto a full W divertor influences the W erosion and subsequently the main plasma W density in ITER-relevant operating scenarios.
- Characterize the effect of transients (edge localized modes and disruptions) on a Be first wall.
- Develop control strategies applicable to ITER for detecting and limiting damage to Be and W plasma-facing components, such as relevant disruption mitigation and edge localized mode (ELM) power loss control systems. The power upgrades allow extension of this work to energy densities in transients comparable to those in ITER.
- Study melt layer behaviour in ELMs and disruptions energy losses and implications for subsequent plasma operation. Possible alloying of Be with W which may reduce the melting point for the Be–W mixed plasma facing near surface layer.
- Develop integrated ITER compatible scenarios for an all-metal machine including impurity seeding strategies to replace the intrinsic carbon radiation which is essential for the achievement of acceptable divertor power loads in the ITER baseline edge scenario.
- Investigate special heating system related effects such as the interaction of fast ions generated by the new JET ITER-like ion cyclotron resonance heating (ICRH) antenna with Be wall components and the W divertor baffle. Recent experiments in the Alcator C-MOD and ASDEX Upgrade tokamaks indicate an increased impurity production with ICRH power and interdependence between antenna performance and plasma edge [5,6].

2.1. Transient ELM and disruption loads on plasma facing components

Under present JET conditions ELMs lead to energy densities on PFCs which can already cause carbon ablation. Taking into account the foreseen input power enhancement in the JET programme (see Section 6), these dangerous regimes will be reached more regularly, calling for adequate control mechanisms to be an integrated part of the future programme. On the other hand, JET will be able to easily access these regimes to deliberately study the effects of such power loads on the target, the plasma and divertor impurity concentration and radiation and on the power handling in subsequent operation. Present experimental evidence indicates that most of the energy ($\geqslant$70%) lost by the main plasma during ELMs is deposited on the divertor target and is toroidally symmetric [7] and that the wetted area for ELM energy deposition is not much increased compared with that between ELMs [8], which for typical JET ELMy H-mode conditions is ~1 m^2, corresponding to a ~5 mm power e-folding length at the outer midplane [9]. The typical timescale for ELM energy deposition at the divertor is of the order of the ion SOL transit time corresponding to the pedestal plasma parameters, covering a range of 100–500 μs in JET. The previous JET high current operation [10,11] indicates that discharges with plasma current $I_p \geqslant 3$ MA and input power $P_{inp} \geqslant 20$ MW can reach Type I ELMy H-modes with ELM energy losses $\Delta W_{ELM} \geqslant 1$ MJ. This is e.g. demonstrated in Fig. 2 where a JET discharge with natural density (no fuelling) is compared to another discharge with additional fuelling (2.5×10^{22} s^{-1}), both with Type I ELMs at $I_p = 3$ MA and $P_{inp} = 20$ MW. In the case of natural density an ELM energy loss of $\Delta W_{ELM} \sim 1$ MJ is reached, which is above the carbon ablation limit. At low pedestal collisionalities $\nu_{ped} \sim 0.05$ (typical of high I_p and P_{inp} JET operation and of the ITER reference regime) a simple scaling of the pedestal energy with I_p^2 [12] would result in an expected ELM energy at $I_p = 5$ MA and $P_{inp} = 40$ MW of $\Delta W_{ELM} \geqslant 2$ MJ with a timescale ~100 μs at an expected pedestal temperature $T_{ped} = 4$ keV. This is significantly above the melting and ablation limits for C and W [13]. As such, the input power plays a smaller role in determining the ELM energy loss than the plasma current. Additional heating power (leading to a total $P_{inp} \sim 45$ MW) will be needed to run ELMy H-mode scenarios at high plasma current, thereby providing access to Type I ELM regimes with ITER-equivalent ΔW_{ELM}.

Evidence for enhanced impurity production during large ELM energy losses has been obtained at JET from the enhanced radiation losses associated with ELMs, which are caused by particles sublimed/ablated at the divertor target during the ELM pulse and re-entering the plasma. As can be seen in Fig. 3, these radiative losses suddenly increase if the ELM energy loss exceed about 0.7 MJ, indicating the enhanced impurity production during the ELM.

The effect of large ELMs on the Be first wall is more complicated to estimate due to the more limited range of available data and lower level of understanding of ELM energy loads on the main chamber of tokamaks [14,15]. However, assuming a significant amount of energy (~30%) reaches the PFCs at the main chamber, for ELMs with $\Delta W_{ELM} > 2$ MJ Be melting (due to the low Be melting temperature) is expected to occur, unless the effective area for ELM energy deposition on the main wall is larger than 2 m^2. This effective area remains to be determined more quantitatively, but existing evidence indicates that the ELM energy is deposited on the elements closest to the plasma separatrix at the outer mid-plane [14]. Therefore, the ELM energy flowing outside the divertor is likely to be deposited on the Be main chamber limiters together with the top X-point tiles, depending on the magnetic configuration.

Operation of JET in good confinement H-modes at the highest currents and input powers will bring

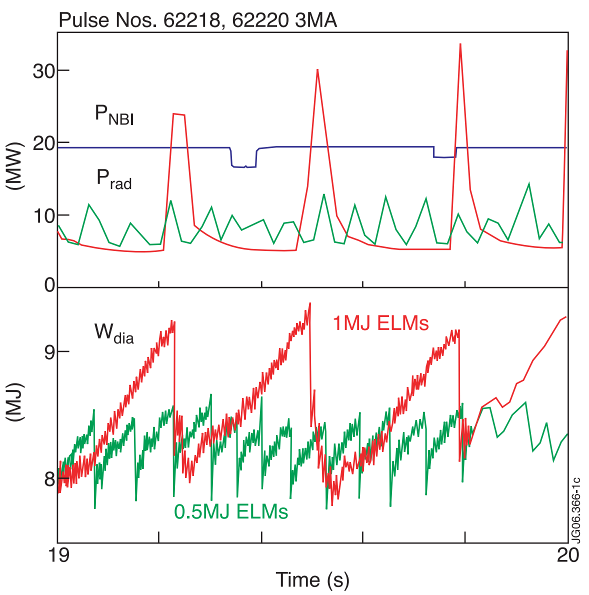

Fig. 2. Two JET 3MA/3T discharges in the DOC-L configuration with Type I ELMs, one with natural density and one with additional fuelling (2.5×10^{22}). In the case of natural density the main plasma ELM energy losses reach ~1 MJ, exceeding the carbon ablation limit.

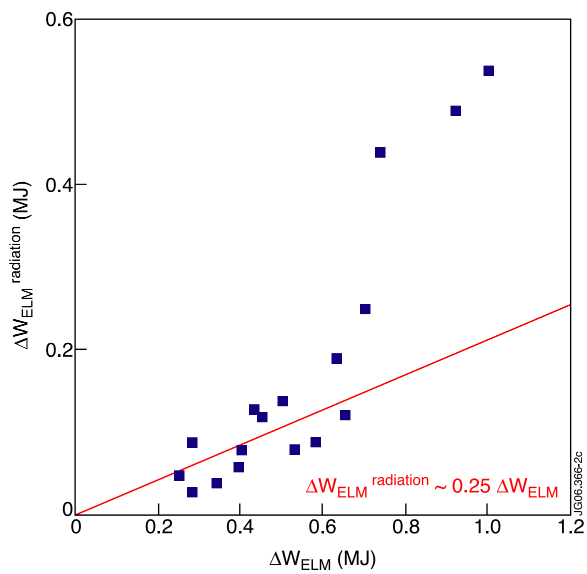

Fig. 3. Radiative losses associated with Type I ELMs in JET versus main plasma ELM energy loss showing enhanced radiation for ~1 MJ ELMs in JET.

the plasma energy to $W_{dia} \sim 20$ MJ. This will allow to study the effect of disruption energy deposition on the PFC expected to occur in ITER. Issues are the possible formation of a vapour shield and the shallow melting of some areas of the first wall caused by disruptions. Mitigation of disruptions in ITER by intense impurity puffing may lead to a radiative energy flux deposited on the Be wall of $\sim$0.5 MJ/m^2 in a short timescale ($\sim$300 μs), which would cause Be-melting to a depth of several tens of μm [16]. The dynamics of such a melt layer and its effect on plasma operation remains a key uncertainty in the operation of ITER and will be addressed in JET. The estimated radiative fluxes on the main Be wall in JET are >0.2 MJ/m^2 in timescales shorter than 100 μs. These loads will bring the main chamber Be wall close to its melting point. Estimations of the energy loads in a worse case disruption in JET [17] lead to an energy flux onto the divertor of $\sim$3–4 MJ/m^2 which would lead to intense carbon evaporation and/or tungsten melting/evaporation. Control mechanism to avoid such conditions must thus be fully developed for the JET wall project, as it will be for ITER, but a careful approach to study the effect of such power loads on melt layer behaviour will be done.

3. Beryllium first wall

3.1. First wall erosion behaviour

A first goal is the identification of the nature and magnitude of the main chamber Be erosion, the ELM and inter-ELM contributions, the possible impact from disruptions and to scale up this behaviour to ITER like conditions. The present estimates for the main chamber interaction and Be erosion in ITER are based on modelling of the inter-ELM behaviour with B2/Eirene, based on a 'classical' SOL model governed by the competition of (classical) parallel transport and assumptions on the (turbulent) perpendicular transport. Erosion is due to charge exchange neutrals and ion sputtering calculated using the modeled plasma parameters at the edge of the B2/Eirene grid. The overall calculated erosion rates are between 6 and 30 g Be/ITER pulse (2×10^{21} Be/s to 1×10^{22} Be/s) [18,1,19]. In the present JET conditions, Be is evaporated routinely on the first wall and Be erosion has been estimated using a combination of spectroscopy and edge modelling to be about 5×10^{19} Be/s on average [20]. An estimate of the average Be coverage of the first wall was done based on a combination of various data yielding a 20% averaged Be coverage of the first wall in JET, scaling up the Be erosion in JET to about 2.6×10^{20} Be/s for a full Be wall. A simple scaling based on the input energy in JET and ITER would scale to a value of about 1×10^{22} Be/s in ITER, similar to values modeled by B2-EIRENE, but other observations indicate a similar range of overall particle flux to the first wall independent of machine size [21], which would also result in similar amounts of eroded Be ($\sim 3 \times 10^{20}$–3×10^{21} Be/s), predicting thus a much lower Be influx in ITER. Presently, no consistent prediction and extrapolation of the first wall interaction in ITER exists and this topic therefore remains a subject of active research.

The level of the interaction of ELMs with the main wall in ITER remains to be quantified. Infra-red measurements in ASDEX Upgrade indicate a maximum of 25% of the ELM energy arriving at plasma facing components in the main chamber. The spatial distribution of the power onto the first wall is not fully diagnosed but present results [14,15] indicate that the energy flow to the main chamber is mostly due to ions with energies typical of the pedestal values reaching convectively the wall during the ELM. This would imply that ions with rather high energies may impact the main wall in ITER during ELMs with important consequences for a high Z first wall (if installed in ITER) and perhaps also for the tungsten divertor baffles in ITER. The effect on Be sputtering in extrapolating from JET to ITER is weaker since Be sputtering yield, largely provided by deuterium impact, is already near its maximum in JET.

3.2. Long range beryllium transport

In general, material redistribution is largely determined by long range impurity transport which is closely linked to the physics of flows and drifts in the SOL being subject of intense research at present. Understanding of the physics basis behind the SOL flows is required for reliable extrapolation of the observed Be migration from JET to ITER. All present divertor experimental devices show the inner divertor to be deposition dominated while the outer target is seen to be an erosion or deposition zone depending on the device and operating conditions. The roles of SOL flows and the dependence on divertor conditions (low recycling/high recycling/ detached) remain to be determined. A Be wall in JET together with a non-Be divertor will provide a valuable insight into the migration processes likely to be encountered in ITER. This will enhance both empirical and code based predictive capabilities.

3.3. Effect of beryllium deposition on tungsten and carbon release

Basic solid state data and recent dedicated laboratory experiments [22] show an alloying of Be deposited on top of W to Be_2W starting to form already at low temperatures (400 K) and a thermal sublimation of the excess Be beginning at about 600 K. The Be_2W layer has a reduced melting point of 2520 K. Both effects can have important consequences for W erosion and the short range Be migration inside the divertor. In option 2 of the ITER-like wall project, CFC will be used at the high heat flux areas of the divertor. A main question here is the suppression of carbon chemical erosion by Be deposition, an effect clearly demonstrated in PISCES [23,24]. This reduction of chemical erosion must be explored in detail in ITER-like plasma conditions, in particular under the influence of divertor transient loads caused by ELMs.

A key element is to quantify T retention in re-deposited mixed Be layers and identify its possible removal techniques.

3.4. Power handling with a beryllium first wall and a tungsten divertor

The power handling scenario for high performance shots in ITER and, in particular DEMO, is based on partial detachment to reduce the peak heat loads to the acceptable technical level of $10\ \mathrm{MW\,m^{-2}}$. The present experimental evidence that supports the choice of this divertor regime for ITER relies on carbon being the main radiator except from hydrogen and the existence of carbon release due to chemical erosion. A scenario with no or largely reduced carbon release requires most probably extrinsic impurity puffing. The various interactions among the various impurities and a W divertor should be assessed for ITER operation with the new JET divertor.

4. Full tungsten divertor

The main goals of a full W divertor in JET are to study the

- *Compatibility of a full W divertor with core plasma performance.* The main subject is still to demonstrate the compatibility of the foreseen ITER scenarios with a full W divertor, in particular whether they suffer from tungsten accumulation as a result of W erosion from the divertor region and, if so, to demonstrate techniques to control it. Importantly, these data will be gained under conditions of a main Be wall and possible increased W target erosion due to seeded impunities needed to replace the absent intrinsic carbon radiation in the edge and divertor region.
- *Erosion and local migration of W and the formation of mixed Be–W re-deposited layers.* Analysis of W erosion characteristics, its local migration behavior and its possible co-deposition with Be (O) to form mixed W/Be (O) layers are to be addressed. Their location, quantity and fuel retention characteristics will provide important information to the expected T retention under the proposed back-up material choice conditions in ITER.
- *Melt layer stability and melt damage evolution.* Arguments against W at the high heat flux areas of ITER range from the melting caused by large (uncontrolled) energy losses during ELMs or disruptions to the possible damage (fatigue cracking) due to energy deposition by a large number of ELMs even with energies below the W melting threshold. As discussed in Section 2.1 JET reaches presently just marginally the threshold condition for W melting and carbon ablation which requires the ELM energy release to exceed about 0.7–1 MJ, a regime which can be more easily explored with the foreseen power upgrade. Using a solid W target at the outer strike point, the effects mentioned above can be (carefully) investigated, providing thereby essential information for a possible use of full W targets in ITER. The key topics are melt layer formation and stability, along with the resulting plasma contamination, the behavior of re-solidified layers, the impact on the subsequent plasma operation and the production of dust.
- *Formation of Be–W alloys.* The formation of Be–W alloys due to deposition of Be on W is among the most critical question to be urgently answered for ITER, also for the present ITER start up material choice. The kinetic of this alloy formation depends on the amount of Be deposition, the surface temperature evolution and the temperature controlled processes of Be and W diffusion through solid Be and W and the Be/W alloy

itself, such as Be_2W. A coordinated research including dedicated lab-experiments, PISCES B plasma simulator and the new JET wall experiment is needed to provide solid predictions of this process for ITER.

5. ITER-like wall in JET – technical realisation

5.1. Beryllium main wall

As shown in Fig. 4, the technical implementation of the JET main chamber tiles foresees the use of bulk Be for all the inner and outer wall guard limiters but excluding parts of two inner wall limiters that are hit directly by neutral beam injection (NBI) shine through. Bulk Be is also foreseen for the upper dump plates, the saddle coil protection tiles (upper and lower), the lower hybrid (LH) and the ICRH antenna protections. The NBI shine through protection tiles are made of W-coated CFC but will be 3 cm recessed with respect to the Be guard limiters. In addition, the protection tiles at the upper divertor baffle areas are made from W-coated CFC, not due to power handling reasons but due to the effort required to design solid W tiles and the high cost. A total number of 4404 main wall tiles currently installed in JET will be replaced by 1700 solid Be tiles machined from 4 tonnes of solid Be in addition to the Be coated inconel tiles which will be placed on the inner wall areas between the limiters and the W-coated CFC tiles. Fig. 5 shows an example of the technical realization for an outer poloidal limiter, demonstrating also the

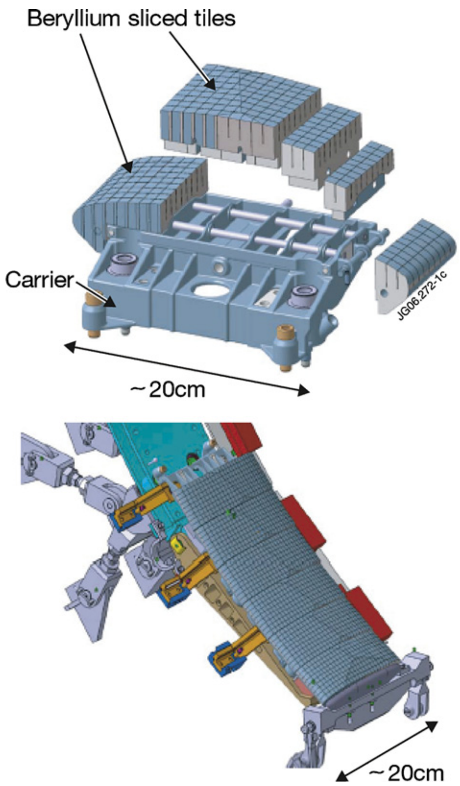

Fig. 5. Example of tile design for the JET main wall structure (here a poloidal limiter).

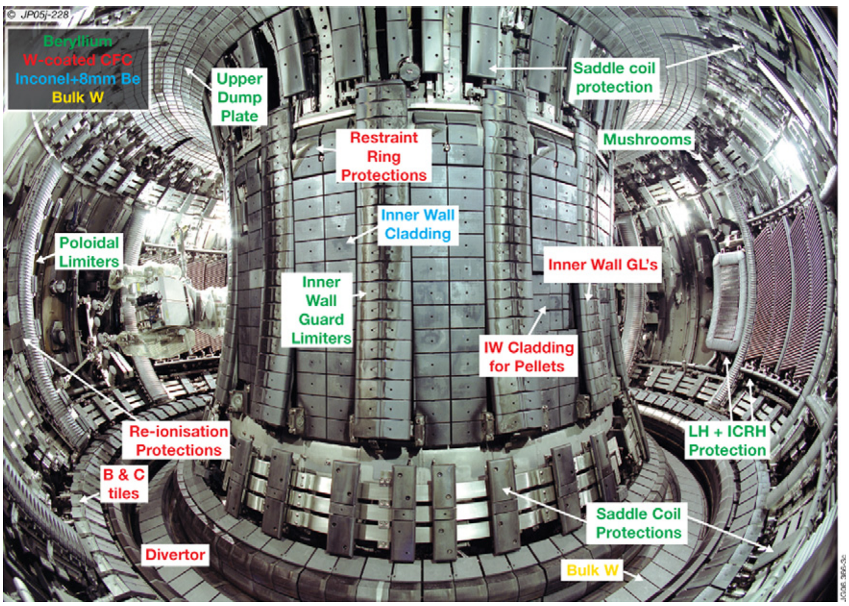

Fig. 4. JET has 4404 main wall tiles to be replaced with beryllium.

considerable engineering effort needed to redesign all main chamber parts. All Be tiles are sliced and castellated toroidally and poloidally in order to reduce electromagnetic forces and increase the heat shock resistance. This design will serve also as an excellent test for ITER in view of the mechanical integrity and the possible material migration into gaps and the associated fuel retention.

5.2. Tungsten divertor/R&D on tungsten coating and bulk tungsten tiles

The technical solution proposed for the JET full W divertor, as shown in Fig. 6, includes bulk W for the load bearing septum replacement plate (LBSRP) and W coating on CFC for the remaining tiles.

In view of the need to select the most reliable W coating technique, a coordinated R&D programme was launched early 2006. As a result, 14 different types of W coatings, including chemical vapour deposition (CVD, 4, 10, 200 μm), physical vapour deposition (PVD, 4, 10 μm) and vacuum plasma spraying (VPS, 200 μm), have been produced and qualified under cyclic heat loading under the coordination of IPP Garching in cooperation with CEA Cadarache, ENEA Frascati, TEKES Finnland, MEdC Romania and FZJ Germany. All coatings were subjected to a thermal screening at the GLADIS test facility (IPP-Garching) with stepwise power loads increasing from 4 MW/m^2 in 6 s up to 22 MW/m^2 in 1.5 s, with the surviving 9 out of 14 coating types exposed afterwards to cycling tests for 200 high heat flux pulses [25]. The different and anisotropic thermal expansion between CFC and W often led to cracking perpendicular to the fibres, inducing delamination by buckling along the fibres, melting and partial loss of coating. A 200 μm VPS coating and a thin 10 μm ion assisted magnetron sputtered layer behaved best. These coatings were additionally exposed to 1000 typical JET like ELMS (0.35 GW/m^2 for 1 ms) in the electron beam facility JUDITH in FZJ, demonstrating their robustness for these number of cycles (note that more than 100 ELMS of this energy can occur in one high performance JET pulse).

For the most heavily power-loaded tiles, a bulk W tile concept has been developed under the leadership of FZJ. The design was mostly determined by the strong constraint of minimising the electromagnetic forces in disruptions and optimising mechanical stability. A design was made of 6 mm thick W lamellae (see Fig. 7), packed together in 4 poloidal stacks and bolted together in toroidal direction but with electrical isolating spacers to reduce eddy currents (see Fig. 8). Each lamella has dedicated electrical contact points to the support structure to reduce halo forces and avoid arcing. A prototype of this concept has survived cycling heat flux tests (200) with 7 MW for 10 s and failure test with 10 MW for 14 s with temperatures exceeding 3000 °C.

6. Accompanying enhancement programme

In order to profit the most from the ITER-like wall experiment in JET and to ensure the coherence

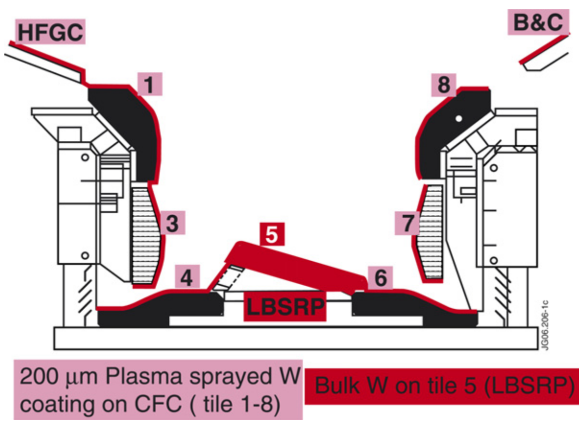

Fig. 6. The JET full W divertor. The load bearing septum replacement plate (LBSRP) is made of bulk tungsten. All other tiles are made of tungsten-coated CFC including the high field gap closure (HFGC) tiles.

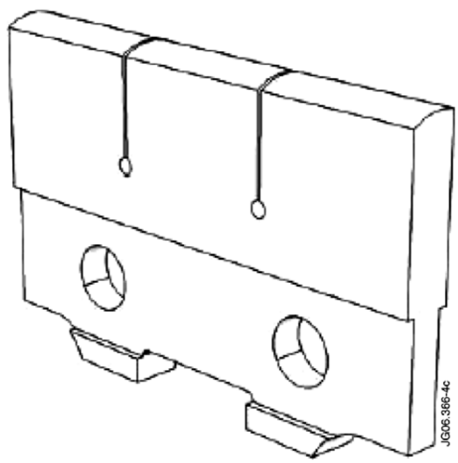

Fig. 7. Typical tungsten lamellae for the stacks assemblies: 6 mm thick, 64 mm long. The height of 40 mm is a result of the expected temperature gradient and of the total heat capacity required, in account for the other dimensions.

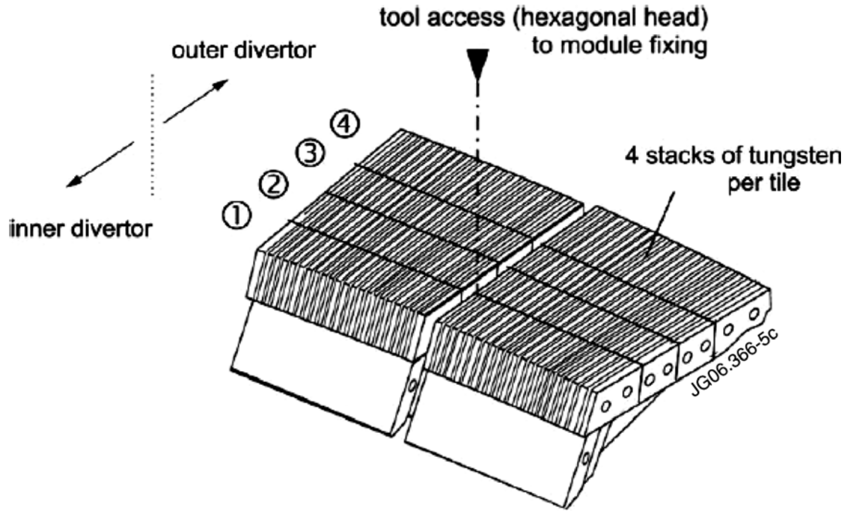

Fig. 8. Arrangement of four stacks for the bulk W tiles. A poloidal span of 60 mm without additional castellation, or deep cuts in poloidal direction, has been considered tolerable if, and only if the lamellae thickness does not exceed a few millimetres. Note that the small tiles on the inner side are made of W-coated CFC.

of the JET programme in support of ITER, a number of enhancements of the JET Facilities are planned. In order to increase the ITER-relevance of the plasma scenarios, additional neutral beam (NB) heating power, up to >34 MW for 20 s (compared to 25 MW for 10 s at present) will be provided by upgrading the existing NB boxes and power supplies for operation at higher current. Phasing the two NB boxes will allow delivering about 17 MW of NB power for up to 40 s, for full exploitation of the pulse length capability of the JET machine. This will be essential to progress, in particular, hybrid and advanced scenarios for ITER, which require full or partial current profile control. Consequently, the overall auxiliary heating power of JET will be increased to ~45 MW, allowing access to ITER-relevant steady-state conditions, disruption and ELM loss power densities, thereby opening up a range of new possibilities for scenario development. With regard to ELMy H-mode scenarios, the work will include stable H-mode operation at 3.4 T and 85% of the Greenwald density as well as pedestal energy and confinement scaling studies. Hybrid scenarios will benefit from the power upgrade thanks to the fact that JET will then be able to operate significantly closer to ITER conditions and provide an improved confinement scaling for extrapolation to ITER. Finally, OD calculations indicate [26] that the increase in power and current drive could dramatically extend the operational space of the advanced tokamak regime in JET in terms of current and density, with high normalized ratio of plasma pressure over magnetic pressure ($\beta_N \sim 3.4$) and bootstrap current fraction in the range of 70% at high toroidal field (3.5 T).

A fast pellet injector (variable pellet speed 50–200 m/s and frequency up to 60 Hz) will be installed in JET to provide means to explore the ELM-pacing technique that is successfully demonstrated on ASDEX-Upgrade [27]. This system is also designed to provide a high fuelling capability (variable pellet volume 35–70 mm^3, pellet speed 100–500 m/s and frequency up to 15 Hz) and will, together with the power upgrade, give access to ELMy H-mode and improved H-modes at higher β_N, under conditions of ELM control.

Most new or upgraded diagnostics are dedicated to support the ITER-like wall experiment. An ITER-relevant (reflective optics) wide angle infra-red viewing system has been installed recently (2005) to investigate plasma–wall interaction in the main chamber. A poloidal array of thermocouples will be supplemented for machine protection and interpretation of infra-red data. A new fast divertor infra-red camera is to be installed in 2007. JET is already equipped with a unique set of samples to detect erosion and deposition in the main chamber and the divertor, such as marker tiles, deposition monitors, rotating collectors and quartz micro-balances (see Fig. 9). These will be re-installed in the new wall experiment to compare the present wall with the new one. Finally, spectroscopic diagnostics

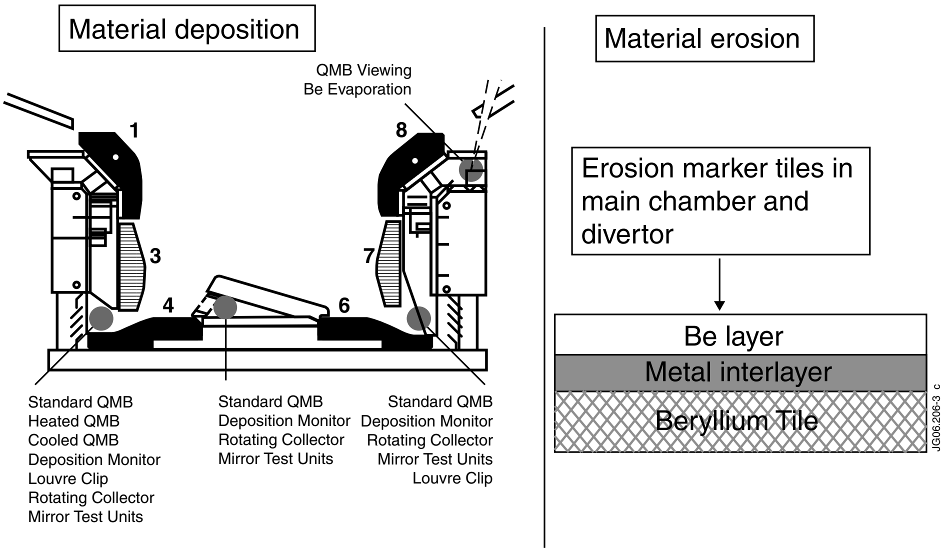

Fig. 9. Examples of erosion and deposition probes in the JET divertor region (left) and main chamber (right). QMB stands for quartz micro-balance.

for impurity sources and core concentrations for Be and W will be improved in the visible, XUV and VUV range.

7. Implications for plasma scenarios

Besides the investigation of a number of dedicated plasma surface interaction questions, the development of ITER-relevant plasma scenarios which are compatible with the new wall and divertor materials mix is the most crucial issue. The melting of Be and W sets much more rigorous power handling limitations as compared with a carbon wall, imposing an accurate control of edge plasma parameters to control power loads in between and during transients. The compatibility of high plasma performance with ELM mitigation schemes will have to be established. Moreover, the present edge conditions during internal transport barrier plasmas represent a constraint for power handling, particle exhaust and W sputtering requirements. A key point will be to develop advanced scenarios with edge conditions satisfying these requirements and compatible with good core confinement [26]. The lower divertor and edge radiation level due to the absence/reduction of carbon will call for impurity seeding, whose impact on plasma performance and W sputtering will need to be investigated. Possible W accumulation in the plasma core is a concern, in particular, in conditions of density peaking expected with ITER-like low collisionalities. Present operational experience with high Z metal PFCs in Alcator C-MOD and ASDEX Upgrade tends to indicate a possible detrimental impact of high Z impurity accumulation on plasma performance [28,29].

8. Summary

JET is underway to completely exchange the present wall materials (CFC) towards a metal dominated ITER-like material mix with mainly solid Be tiles in the main chamber and a full W divertor. This large effort is motivated by a number of outstanding questions for which urgent answers are needed for ITER construction and operation. The most important goals are to

- Demonstrate that a Be wall plus an all-W divertor has sufficiently low fuel retention to meet ITER requirements and, if not successful, to develop appropriate mitigation and removal schemes. It should be noted that T removal in a mixed metal environment is still an open issue.
- Characterize the main plasma wall interaction during, in between ELMs and in disruptions and explore plasma scenarios compatible with a low power handling Be wall which can be subject to melting.
- Study the formation of mixed layers and their impact on erosion and material migration.
- Study melt layer stability and melt damage evolution.

- Develop control strategies for detecting and limiting damage to Be and W plasma-facing components, such as relevant disruption mitigation systems and ELM power loss control.
- Develop integrated scenarios for an all-metal machine including impurity seeding strategies to replace the intrinsic carbon radiation to achieve acceptable divertor power loads in the current ITER baseline edge scenario.
- Demonstrate the compatibility of the foreseen ITER scenarios with a full W divertor and Be wall.

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